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Waterford 3

W3F1-2002-0107

December 20, 2002

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

SUBJECT: Waterford Steam Electric Station, Unit 3
Docket 50-382
License Amendment Request NPF-38-244
Corrections and Clarifications to Various Technical Specification Pages

Dear Sir or Madam:

Pursuant to 10 CFR 50.90, Entergy Operations, Inc. (Entergy) hereby requests the following amendment for Waterford Steam Electric Station, Unit 3 (Waterford 3). Entergy requests that several administrative changes be made to revise, delete, correct or clarify certain titles, page numbers, and heading information. Entergy also proposes to revise personnel and committee titles that have been changed and revise administrative reporting requirements to conform to 10 CFR 50.4. Additionally, Entergy proposes to delete redundant or unnecessary requirements from Technical Specifications 5.4, "Reactor Coolant System," Technical Specification 6.6, "Reportable Event Action," and Technical Specification 6 7, "Safety Limit Violation" consistent with NUREG 1432, Revision 2.

Several index pages being changed by this license amendment request are also impacted by other license amendment requests currently under NRC review. When replacement pages are requested, Entergy will provide Technical Specification pages reflecting the latest approved version

The proposed amendment has been evaluated in accordance with 10 CFR 50.91(a)(1) using criteria in 10 CFR 50.92(c) and it has been determined that this change involves no significant hazards consideration The basis for this determination is included in Attachment 1.

The proposed changes do not include any new commitments. Entergy requests approval of the proposed amendment within one year of the date of this letter. Once approved, the amendment shall be implemented within 60 days.

A001

If you have any questions or require additional information, please contact Jerry Burford at extension 601-368-5755.

I declare under penalty of perjury that the foregoing is true and correct. Executed on December 20, 2002.

Sincerely,

J. E. Venable
Vice President, Operations
Waterford Steam Electric Station, Unit 3

JEV/LAE/cbh

Attachments:
1. Analysis of Proposed Technical Specification Change
2. Proposed Technical Specification Changes (mark-up)
3. Changes to Technical Specifications Bases Pages (Information Only)

cc: Mr. Ellis W. Merschoff, NRC Regin IV
Mr. N. Kalyanam
NRC Resident Inspectors Office
J. Smith (Wise Carter)
N. S. Reynolds
Louisiana DEQ / Surveillance Section
American Nuclear Insurers

Attachment 1

W3F1-2002-0107

Analysis of Proposed Technical Specification Change

1.0 DESCRIPTION

This letter is a request to amend Operating License NPF-38 for Waterford Steam Electric Station, Unit 3 (Waterford 3).

The proposed change will revise the Technical Specifications (TS) to correct, clarify or delete information, and/or redundant requirements. Most of the changes are administrative in nature however several minor technical changes are also proposed.

2.0 PROPOSED CHANGES

Entergy proposes the following changes to the Waterford 3 TS:

- INDEX page V: add "–Fxy" to PLANAR RADIAL PEAKING FACTORS title and "-Tq" to AZIMUTHAL POWER TILT title to match titles on pages 3/4 2-3 and 3/4 2-4 respectively.

- INDEX page VI: indent subsections under 3/4.4.1 to match format on page and add "SPRAY" to AUXILIARY title under 3/4.4.3 to match the title on page 3/4 4-9a, which was added in Amendment 22.

- INDEX page VII: add "-W" to ELECTRIC HYDROGEN RECOMBINERS title to match the title on page 3/4 6-35.

- INDEX page VIII: add subsection identification for TS 3/4.7.6 and indent 3/4.8.2 D.C.SOURCES to match page format. The insert adds the following subsections: CONTROL ROOM EMERGENCY AIR FILTRATION SYSTEM - OPERATING; CONTROL ROOM EMERGENCY AIR FILTRATION SYSTEM - SHUTDOWN; CONTROL ROOM AIR TEMPERATURE - OPERATING; CONTROL ROOM AIR TEMPERATURE - SHUTDOWN, and CONTROL ROOM ISOLATION AND PRESSURIZATION with corresponding page numbers. These index additions reflect changes made in Amendment 115.

- INDEX page XI: add "-Fxy" to PLANAR RADIAL PEAKING FACTORS title to match title on page B 3/4 2-2; add "-Tq" to AZIMUTHAL POWER TILT title to match title on page B 3/4 2-2; correct page number for MONITORING INSTRUMENTATION from B 3/4 3-1 to B 3/4 3-2; and delete 3/4.3.4 TURBINE OVERSPEED PROTECTION reference as there is no place holder for this bases section, which was deleted in Amendment 103.

- INDEX page XII: revise page number for ECCS SUBSYSTEMS to read B 3/4 5-1b; revise page number for REFUELING WATER STORAGE POOL to read B 3/4 5-3; revise page number for CONTAINMENT ISOLATION VALVES to read B 3/4 6-5; revise page number for COMBUSTIBLE GAS CONTROL to read B 3/4 6-6; revise page number for VACUUM RELIEF VALVES to read B 3/4 6-6; revise page number for SECONDARY CONTAINMENT to read B 3/4 6-7; and revise page number for STEAM GENERATOR PRESSURE/ TEMPERATURE LIMITATION to read B 3/4 7-3b.

- INDEX page XIII: revise page number for COMPONENT COOLING WATER AND AUXILIARY COMPONENT COOLING WATER SYSTEMS to read B 3/4 7-3b; capitalize "and" to be consistent with title on page; revise page number for FLOOD PROTECTION to read B 3/4 7-4a; revise page number for CONTROL ROOM AIR CONDITIONING SYSTEM to read B 3/4 7-4a; revise page number for CONTROLLED VENTILATION AREA SYSTEM to read B 3/4 7-4c; and revise page number for ESSENTIAL SERVICES CHILLED WATER SYSTEM to read B 3/4 7-7.

- INDEX page XIV: delete "3/4.9.12 DELETED.....B 3/4 9-3" as there is no place holder for this section, which was deleted in Amendment 176.

- INDEX page XV: delete section 5.4 title REACTOR COOLANT SYSTEM and insert "Not Used", delete subsection 5.4.1 and subsection 5.4.2 to agree with requested change to page 5-5 below. Add an "S" to LIMIT in the COMPONENT CYCLIC OR TRANSIENT LIMIT title to match the title on page 5-6.

- INDEX page XVI: add "AND ONSITE ORGANIZATIONS" after OFFSITE in the title for SECTION 6.2.1 to match the title on page 6-1; delete titles, subtitles and subtitle page numbers after SECTION 6.2.3 and insert "Not Used" to agree with changes made in Amendment 146; delete REVIEW AND AUDIT after SECTION 6.5 and replace with "Not Used" and delete SECTION 6.5.1 in its entirety to agree with changes made in Amendment 109. These changes match the index with the cited TS page and previously approved changes.

- INDEX page XVII: delete SECTION 6.5.2 in its entirety to agree with changes made in Amendment 109; delete section 6.6 title REPORTABLE EVENT ACTION and page number and insert "Not Used" to agree with changes proposed in this letter; delete section 6.7 title SAFETY LIMIT VIOLATION and page number and insert "Not Used" to agree with changes proposed in this letter; revise page number for REPORTING REQUIREMENTS to read 6-17; revise page number for ROUTINE REPORTS to read 6-17; insert "COLR" after CORE OPERATING LIMITS REPORT; and delete RECORD RETENTION and page number after SECTION 6.10 and replace with "Not Used" to agree with changes made in Amendment 146.

- INDEX page XVIII: add "(PCP)" after PROCESS CONTROL PROGRAM and "(ODCM)" after OFFSITE DOSE CALCULATION MANUAL to match titles on pages 6-23 and 6-24, respectively.

- INDEX page XIX: change PRESSURE/TEMPERATURE LIMITATIONS FOR 0-8 EFFECTIVE FULL POWER YEARS to "PRESSURE-TEMPERATURE LIMITS 0-16 EFPY" after REACTOR COOLANT SYSTEM in titles to figures 3.4-2 and 3.4-3 to match titles of corresponding figures on pages 3/4 4-30 and 4-31 that were revised in Amendment 160; add FIGURE "3.6-1 DELETED 3/4 6-12" to reflect the place holder for the figure that was deleted in Amendment 27; insert titles for Figures 5.6-1, 5.6-2, and 5.6-3 with respective page numbers after FIGURE 5.1-3 to reflect the figures added by Amendment 144; and in FIGURES 6.2-1 and 6.2-2 replace the titles with words "DELETED" to reflect the place holders for the tables deleted by Amendment 41.

- INDEX page XX: delete Table 2.2-2 titles and page number to reflect the disposition of the table, which was deleted in Amendment 5.

- INDEX page XXI: delete Table 3.3-12 and 4.3-8, the words "DELETED" and associated page numbers as there are no longer any place holders for these tables, which were deleted in Amendment 68.

- INDEX page XXII: add "PER TRAIN" after ULTIMATE HEAT SINK FAN REQUIREMENTS; insert TABLE "4.7-2 SNUBBER VISUAL INSPECTION INTERVAL....3/4 7-21a" after TABLE 3.7-3 to reflect the table added by Amendment 73; delete information related to Tables 4.8-1 and 4.8-1a, which were deleted per Amendment 126.

- Page 2-1: delete ", and comply with the requirements of Specification 6.7.1" in four ACTION statements.

- Page 3/4 3-4a: ACTION 2 change Plant Operations Review Committee to "On-site Safety Review Committee" to reflect correct title.

- Page 3/4 3-17: ACTION 13 change Plant Operations Review to "On-site Safety Review" to reflect correct title.

- Page 3/4 3-18a: ACTION 19a change Plant Operations to "On-site Safety" to reflect correct title.

- Page 3/4 4-32: delete information on this page and insert "THIS PAGE NOT USED" for consistent format; also, rotate the page to portrait format.

- Page 3/4 7-16: change title after <u>PLANT SYSTEMS</u> to read "<u>3/4.7.6 CONTROL ROOM AIR CONDITIONING SYSTEM</u>" and add "CONTROL ROOM EMERGENCY AIR FILTRATION SYSTEM – OPERATING" before LIMITING CONDITION FOR OPERATION to better conform to the Technical Specification format.

- Page 3/4 7-18: delete 3/4.7.6.2 above LIMITING CONDITION FOR OPERATION and change title to read "CONTROL ROOM EMERGENCY AIR FILTRATION SYSTEM – SHUTDOWN" to better conform to the Technical Specification format.

- Page 3/4 7-18a: delete 3/4.7.6.3 above LIMITING CONDITION FOR OPERATION and change title to read "CONTROL ROOM AIR TEMPERATURE - OPERATING" to better conform to the Technical Specification format.

- Page 3/4 7-18b: delete 3/4.7.6.4 above LIMITING CONDITION FOR OPERATION and change title to read "CONTROL ROOM AIR TEMPERATURE – SHUTDOWN" to better conform to the Technical Specification format.

- Page 3/4 7-18c: delete 3/4.7.6.5 above LIMITING CONDITION FOR OPERATION to better conform to the Technical Specification format.

- Page 3/4 8-7: delete information on this page and insert "THIS PAGE NOT USED" for consistent format.

- Page 3/4 11-18: delete page in its entirety as it is at the end of the section and is not needed as a place holder.
- Page 5-5: delete section 5.4 title REACTOR COOLANT SYSTEM and insert "Not Used"; delete section 5.4.1 DESIGN PRESSURE AND TEMPERATURE, and section 5.4.2 VOLUME in their entirety to conform to NUREG 1432, Revision 2 "Standard Technical Specifications – Combustion Engineering Plants" which no longer includes this information in the Design Features section.
- Page 5-6: correct spelling of refuelling to "refueling" in section 5.6.1.c and correct spelling of accceptable to "acceptable" in section 5.6.1.f.
- Page 6-2a ADMINISTRATIVE CONTROLS paragraph f.: change Operations Superintendent to "Assistant Operations Manager (shift)" to reflect correct title.
- Page 6-5 TABLE 6.2-1: change Superintendent to "Manager" (four places) and the acronym SS to "SM" (three places) to correct titles.
- Page 6-6 section 6.2.4.1: change Superintendent to "Manager" to correct title.
- Page 6-8: change 6-12 to read "6-13" in the middle of the page and 6-13 to "6-14" at the bottom of the page to reflect the deletion of page 6-13 as proposed in the next item below.
- Page 6-13 and 6-14: delete TSs 6.6 and 6.7 and delete page 6-13 in its entirety.
- Page 6-14: insert "6.6 Not Used" and "6.7 Not Used" above 6.8 and change PORC to "On-site Safety Review Committee" in NOTES: (1).
- Page 6-17: revise TS 6.9.1 to read, "The following reports shall be submitted in accordance with 10CFR 50.4."
- Page 6-18: revise TS 6.9.1.6 by deleting "to the Director, Office of Resource Management, U. S. Nuclear Regulatory Commission, Washington, DC 20555, with a copy to the Regional Administrator of the regional Office of the NRC."
- Page 6-20a: revise TS 6.9.2 to read "Special Reports shall be submitted in accordance with 10 CFR 50.4 within the time period specified for each report."

The following changes to the TS Bases pages are to be made by Waterford 3 in accordance with the TS Bases Control Program. They are presented here for information only. Marked up pages are presented in Attachment 3.

- Page B 3/4 2-2: add "–Fxy" to PLANAR RADIAL PEAKING FACTORS to reflect the technical specification.
- Page B 3/4 3-3 BASES section 3/4.3.3.6 ACCIDENT MONITORING INSTRUMENTATION: change PORC to "OSRC".

- Page B 3/4 7-4a: insert "3/4.7 6 CONTROL ROOM AIR CONDITIONER SYSTEM" above 3/4.7.6 1 and 3/4.7.6.2 CONTROL ROOM EMERGENCY AIR FILTRATION SYSTEM.

- Page B 3/4 7-5 BASES 3/4.7.8 change Plant Operations to "On-site Safety".

In summary, most of the proposed changes are administrative in nature and maintain compliance with the regulatory requirements Several minor technical changes are also proposed and justified below.

3.0 BACKGROUND

This submittal contains a number of administrative changes to the Index pages to conform them to approved pages or pages with requested similar title changes in the corresponding TS text. These changes resulted from an administrative review to correct such minor index discrepancies. Changes are also proposed to delete Section 5.4 REACTOR COOLANT SYSTEM, Section 6.6 "REPORTABLE EVENT ACTION" and Section 6.7, "SAFETY LIMIT VIOLATION," and to revise recently changed personnel and committee titles.

A review of the TS determined that the changes to the index pages were not consistently requested when changes were made to the specification pages. The index pages and selected TS pages listed in Section 2.0 above are being changed because the titles in the index do not match the corresponding titles on the approved TS page, or to correct formatting (indenting) or to correct page numbers where TS pages were changed in previous amendments. These discrepancies are being corrected. These changes are strictly administrative in nature and have no impact on the operation of the plant.

A concern regarding potentially misleading information in the Reactor Coolant System (RCS) volume discussion in the Design Features section of the TSs was documented in the Waterford 3 Corrective Action Program. The RCS volume (11,800 cubic feet) contained in TS 5.4.2 is consistent with the nominal design value listed in Final Safety Analysis Report (FSAR) Table 5.1-2 and Table 5.1-3 and with other references throughout the FSAR and is correct. Prior to the T-hot reduction change (approved in Amendment 120) FSAR Table 5.1-2 listed the average operating temperature (T avg) as 582.1°F, which matches the temperature listed in TS 5.4.2. However, following the T-hot reduction change this number was revised to 574° F in the FSAR table but not in the TS. The temperature in TS 5.4.2 is a nominal temperature at which the RCS volume was originally calculated. The approximately 8°F change has a negligible effect on the RCS volume. Since the nominal temperature is not a controlling temperature for plant operation, the current nominal value Tavg. was determined to be adequate until such time as the TS was amended to delete TS 5.4. The deletion of these sections is consistent with NUREG 1432, Revision 2.

TS 6.6 REPORTABLE EVENT ACTION and TS 6.7 SAFETY LIMIT VIOLATION are being deleted because they duplicate the reporting requirements of 10 CFR 50.72, 50.73, and 50.36. This change is also consistent with NUREG-1432, Revision 2.

TS 6.9.1 and 6.9.2 are being revised to conform the TS to an NRC rule change (51 FR 40303) dated November 6, 1986 which standardized, to the extent practical, administrative reporting requirements.

Several pages are being revised to correct obsolete position or committee titles. These changes are a result of internal efforts to standardize certain position or committee titles within Entergy. Specifically, "Operations Superintendent" is being changed to "Assistant Operations Manager (shift)" and "Shift Superintendent" is being changed to "Shift Manager." The duties remain unchanged In addition, the Plant Operations Review Committee title has been changed to the On-site Safety Review Committee.

4.0 TECHNICAL ANALYSIS

4.1 Technical Specification Index and Selected Page Changes

These changes are strictly administrative to conform titles and page numbers or involve minor reformatting without affecting technical content or operational requirements and have no impact on the operation of the plant.

4.2 Technical Specification 5.4

TS 5.4.1 and TS 5.4.2 are being deleted because they do not meet the criteria of 10CFR 50.36 (c)(4). Design pressure and temperature are described in Section 5.2 of the FSAR, making this section redundant. This proposal is consistent with NUREG-1432, Revision 2, "Improved Standard Technical Specifications, Combustion Engineering Plants," which no longer includes RCS design temperature and pressure in the Design Features section. Deleting this section from the Waterford 3 TS creates no operational concerns.

TS 5.4.2 provides a total water and steam volume for the RCS at a nominal Tavg. RCS operating temperature is controlled under 10CFR50.36(c)(2), Limiting Conditions for Operation. TS LCOs 3.1.1.4, "Minimum Temperature for Criticality," and 3.2.6, "Reactor Coolant Cold Leg Temperature," present requirements for the RCS temperature for plant operation and remain unchanged. In accordance with 10CFR50.36(c)(4), design features to be included in the TS are those features of the facility such as materials of construction and geometric arrangements, which if altered or modified would have a significant effect on safety and are not covered in paragraphs (c) (1), (2) and (3) of this section. RCS volume does not meet the criteria of 10CFR50.36; on this basis Entergy proposes this parameter be deleted. This proposal is consistent with NUREG 1432, Revision 2, which no longer includes RCS volume in the Design Features section. Deleting this section from the Waterford 3 TS creates no operational concerns.

4.3 Technical Specifications 6.6 and 6.7

The requirements for notification of reportable events are clearly defined in 10 CFR 50.72 and 50.73. The requirement of TS 6.6 is redundant to these regulations and is not required and is proposed to be deleted.

TS 6.7.1 contains actions to be taken in the event of a Safety Limit violation. These actions require the NRC Operations Center to be notified within one hour, a report be prepared and submitted to the Commission, and operation not to be resumed until authorized by the Commission. Specification 6.7.1 denotes details relating to Safety Limit violations. These details (1) require notification to the Site Vice President and the SRC within 24 hours following violation of a Safety Limit, (2) specify the contents of the Safety Limit Violation Report, and

(3) require review of the Safety Limit Violation Report by OSRC, SRC, and the Site Vice President within 14 days of the violation. These requirements are after-the-fact notifications and reviews and are not necessary to assure safe operation of the facility. As such, this requirement is not necessary to be included in the Technical Specifications to provide adequate protection of the public health and safety. These requirements are duplicated in 10 CFR 50.36, 10 CFR 50.72, and 10 CFR 50.73 and do not need to be repeated in the TS. Since Waterford 3 is still required to meet these requirements outside of current TS control, their removal is considered to be administrative.

Because TS Safety Limit Action 2.1.1.1, Safety Limit Action 2.1.1.2, and Safety Limit Action 2.1.2 reference TS 6.7.1, a conforming change is made to page 2-1. In addition to the justifications provided above, the changes are consistent with NUREG-1432, as modified by TSTF-5, Revision 1.

4.4 Technical Specification 6.9.1

On November 6, 1986 the NRC amended its regulations (51 FR 40303) to establish procedures for submitting correspondence, reports, applications, or other written communications pertaining to the domestic licensing of production and utilization facilities. The rule superseded all existing requirements and guidance with respect to the number of copies and mailing procedures. Licensees whose technical specifications contained conflicting submittal directions were authorized by the rule to delete conflicting directions by pen-and-ink changes to their technical specifications. The Commission did not require formal applications for amendments to effect these changes. The changes proposed to TS 6.9.1, TS 6.9.1.6, and TS 6.9.2 formally conform the current Waterford 3 TS to this rule change by deleting superceded information and directing that the reports be submitted in accordance with 10 CFR 50.4 within the time period specified for each report.

5.0 REGULATORY ANALYSIS

5.1 Applicable Regulatory Requirements/Criteria

The proposed changes have been evaluated to determine whether applicable regulations and requirements continue to be met. The proposed changes to the index pages, associated Technical Specifications (TS) page headings and page numbers, titles and reporting requirements are editorial and maintain compliance with the intent of regulatory requirements. The proposed change to delete the design features related to the reactor coolant system design pressure, temperature, and volume (TS 5.4) does not revise any design or operating parameters and is consistent with the regulatory criteria of 10CFR50.36 (c)(4).

Entergy has determined that the proposed changes do not require any exemptions or relief from regulatory requirements, other than the TS, and do not affect conformance with any General Design Criteria differently than described in the Final Safety Analysis Report (FSAR).

5.2 No Significant Hazards Consideration

Entergy Operations, Inc. (Entergy) proposes to amend the Waterford Steam Electric Station, Unit 3 (Waterford 3) Technical Specifications (TS) to correct or clarify certain information. The proposed amendment will make changes to the index pages, associated TS page headings and page numbers, certain position and committee titles, and reporting requirements and will delete

TS 5.4 "Reactor Coolant System", TS 6.6 "Reportable Event Action", and TS 6.7 "Safety Limit Violation."

Entergy has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No.

   The proposed changes are primarily to correct titles, page numbering errors, and otherwise make the TS index pages consistent with other NRC approved pages. These changes are all of an administrative nature and have no effect on any plant equipment or structures. Therefore, these changes do not increase the probability or consequences of an accident previously evaluated.

   The proposed amendment also deletes TS 5.4.1 and 5.4.2. Values for RCS design pressure, temperature, and volume are contained in the Final Safety Analysis Report. Any changes to these are controlled by 10 CFR 50.59. Therefore, removing the section from the TS will not increase the probability or consequences of previously evaluated accidents.

   The proposed amendment also deletes TS 6.6 and 6.7, and revises TS 6.9.1 and TS 6.9.2 to administratively conform reporting requirements to those in 10 CFR 50. Therefore, removing or revising these sections from the TS will not increase the probability or consequences of previously evaluated accidents.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No.

   The proposed changes are administrative in nature and do not involve a physical alteration of the plant. No new or different equipment or modes of operation are being introduced by this proposed change. Thus, the changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

   Response: No.

   Margin of safety is related to the confidence in the ability of the fission product barriers to perform their design functions during and following an accident situation. These barriers include the fuel cladding, the reactor coolant system, and the containment system. The proposed changes are primarily administrative in nature and can not affect any safety barriers. The proposed change to TS 5.4 only deletes unnecessary information.

Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, Entergy concludes that the proposed amendment(s) present no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

5.3 Environmental Considerations

The proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

Attachment 2

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Proposed Technical Specification Changes (mark-up)

INDEX

LIMITING CONDITION FOR OPERATION AND SURVEILLANCE REQUIREMENTS

INDEX

LIMITING CONDITION FOR OPERATION AND SURVEILLANCE REQUIREMENTS

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

Insert 3/4 7.6

INDEX

BASES

INDEX

BASES

SECTION PAGE

3/4.4 REACTOR COOLANT SYSTEM (Continued)

3/4.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3/4.6 CONTAINMENT SYSTEMS

3/4.7 PLANT SYSTEMS

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BASES

INDEX

BASES

INDEX

DESIGN FEATURES

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ADMINISTRATIVE CONTROLS

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ADMINISTRATIVE CONTROLS

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ADMINISTRATIVE CONTROLS

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LIST OF FIGURES

Insert Figures 5.6 Here

Insert Figures 5.6

INDEX

LIST OF TABLES

INDEX

LIST OF TABLES (Continued)

INDEX

LIST OF TABLES (Continued)

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

2.1.1 REACTOR CORE

DNBR

2.1.1.1 The DNBR of the reactor core shall be maintained greater than or equal to 1.26.

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the DNBR of the reactor has decreased to less than 1.26, be in HOT STANDBY within 1 hour~~, and comply with the requirements of Specification 6.7.1~~.

PEAK FUEL CENTERLINE TEMPERATURE

2.1.1.2 The peak fuel centerline temperature shall be maintained less than 5080°F (decreasing by 58°F per 10,000 MWD/MTU for burnup and adjusting for burnable poisons per CENPD-382-P-A.)

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the peak fuel centerline temperature has equaled or exceeded 5080°F (decreasing by 58°F per 10,000 MWD/MTU for burnup and adjusting for burnable poisons per CENPD-382-P-A), be in HOT STANDBY within 1 hour~~, and comply with the requirements of Specification 6.7.1~~.

REACTOR COOLANT SYSTEM PRESSURE

2.1.2 The Reactor Coolant System pressure shall not exceed 2750 psia.

APPLICABILITY: MODES 1, 2, 3, 4, and 5.

ACTION:

MODES 1 and 2

Whenever the Reactor Coolant System pressure has exceeded 2750 psia, be in HOT STANDBY with the Reactor Coolant System pressure within its limit within 1 hour~~, and comply with the requirements of Specification 6.7.1~~.

MODES 3, 4, and 5

Whenever the Reactor Coolant System pressure has exceeded 2750 psia, reduce the Reactor Coolant System pressure to within its limit within 5 minutes~~, and comply with the requirements of Specification 6.7.1~~.

TABLE 3.3-1 (Continued)

TABLE NOTATION

ACTION STATEMENTS

ACTION 1 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours and/or open the protective system trip breakers.

ACTION 2 - With the number of channels OPERABLE one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may continue provided the inoperable channel is placed in the bypassed or tripped condition within 1 hour. If the inoperable channel is bypassed, the desirability of maintaining this channel in the bypassed condition shall be documented by the ~~Plant Operations Review Committee~~ in accordance with plant administrative procedures. The channel shall be returned to OPERABLE status prior to STARTUP following the next COLD SHUTDOWN.

On-site Safety Review Committee

TABLE 3.3-3 (Continued)

TABLE NOTATION

(a) Trip function may be bypassed in this MODE when pressurizer pressure is less than 400 psia; bypass shall be automatically removed when pressurizer pressure is greater than or equal to 500 psia.

(b) An SIAS signal is first necessary to enable CSAS logic.

* The provisions of Specification 3.0.4 are not applicable.

ACTION STATEMENTS

ACTION 12 - With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

ACTION 13 - With the number of channels OPERABLE one less than the Total Number of Channels, STARTUP and/or POWER OPERATION and/or operation in the other applicable MODE(S) may continue provided the inoperable channel is placed in the bypassed or tripped condition within 1 hour. If the inoperable channel is bypassed, the desirability of maintaining this channel in the bypassed condition shall be documented by the ~~Plant Operations Review~~ Committee in accordance with plant administrative procedures The channel shall be returned to OPERABLE status no later than prior to entry into the applicable MODE(S) following the next COLD SHUTDOWN. On-site Safety Review

With a channel process measurement circuit that affects multiple functional units inoperable or in test, bypass or trip all associated functional units as listed below:

	Process Measurement Circuit	Functional Unit Bypassed/Tripped
1.	Containment Pressure - High	Containment Pressure - High (ESF) Containment Pressure - High (RPS)
2.	Steam Generator Pressure - Low	Steam Generator Pressure - Low Steam Generator ΔP 1 and 2 (EFAS)

TABLE 3.3-3 (Continued)

TABLE NOTATION

ACTION 17 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may continue provided the inoperable channel is placed in the tripped (D C. Relay energized) condition within 1 hour, the remaining Emergency Diesel Generator is OPERABLE, and the inoperable channel is restored to OPERABLE status within the next 48 hours. Otherwise, be in at least HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the next 30 hours. The surveillance requirements of Table 4.3-2 are waived for all channels while this action requirement is in effect.

ACTION 18 - With more than one channel inoperable, or if the inoperable channel cannot be placed in the trip (D.C. Relay energized) condition, declare the associated Emergency Diesel Generator inoperable and take the ACTION required by Specification 3.8.1.1. The surveillance requirements of Table 4.3-2 are waived for all channels while this action requirement is in effect.

ACTION 19 - With the number of channels OPERABLE one less than the Total Number of Channels, STARTUP and/or POWER OPERATION and/or operation in the other applicable MODE(S) may continue, provided the inoperable channel is placed in the bypassed ~~or tripped condition~~ within 1 hour:

a. If the inoperable channel is to remain in the bypassed condition, the desirability of maintaining this channel in the bypassed condition shall be documented by the ~~Plant Operations~~ On-site Safety Review Committee in accordance with plant administrative procedures. The channel shall be returned to OPERABLE status no later than prior to entry into the applicable MODE(S) following the next COLD SHUTDOWN.

b. If the inoperable channel is required to be placed in the tripped condition, within 48 hours either restore the channel to OPERABLE status or place the channel in the bypassed condition. If the tripped channel can not be returned to OPERABLE status in 48 hours, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours or place the tripped channel in bypass.

With a channel process measurement circuit that affects multiple functional units inoperable or in test, bypass or trip all associated functional units as listed below:

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TABLE ~~4.4.5~~

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PLANT SYSTEMS

AIR CONDITIONING

3/4 7.6.~~1~~ CONTROL ROOM ~~EMERGENCY AIR FILTRATION~~ SYSTEM

CONTROL ROOM EMERGENCY AIR FILTRATION SYSTEM - OPERATING

LIMITING CONDITION FOR OPERATION

3 7.6.1 Both control room emergency air filtration trains (S-8) shall be OPERABLE.

APPLICABILITY*: MODES 1, 2, 3, and 4.

ACTION:

a. With one control room emergency air filtration train inoperable, either restore the inoperable train to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With both control room emergency air filtration trains inoperable, restore one train to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.6 1 Each control room air filtration train (S-8) shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 10 continuous hours with the heaters on.

b. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire or chemical release in any ventilation zone communicating with the system by:

1. Verifying that the filtration train satisfies the in-place testing acceptance criteria and uses the test procedures of Regulatory Positions C.5.a, C.5.c, and C.5 d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is 4225 cfm ±10%.

*During movement of irradiated fuel assemblies, TS 3.7.6.2 is also applicable.

PLANT SYSTEMS

3/4.7.6.2 CONTROL ROOM EMERGENCY AIR FILTRATION SYSTEM - SHUT DOWN

LIMITING CONDITION FOR OPERATION

3.7.6.2 Two control room emergency air filtration trains (S-8) shall be OPERABLE.

APPLICABILITY: MODES 5, 6, and during movement of irradiated fuel assemblies.

ACTION:

a. With one control room emergency air filtration system inoperable, restore the inoperable system to OPERABLE status within 7 days or initiate and maintain operation of the remaining OPERABLE control room emergency air filtration system in the recirculation mode.

b. With both control room emergency air filtration systems inoperable, or with the OPERABLE control room emergency air filtration system, required to be in the recirculation mode by ACTION a, not capable of being powered by an OPERABLE emergency power source, suspend all operations involving CORE ALTERATIONS and movement of irradiated fuel assemblies.

SURVEILLANCE REQUIREMENTS

4.7.6.2 The control room emergency air filtration trains (S-8) shall be demonstrated OPERABLE per the applicable Surveillance Requirements of 4.7.6.1.

PLANT SYSTEMS

~~3/4.7.6.3~~ CONTROL ROOM AIR TEMPERATURE - OPERATING

LIMITING CONDITION FOR OPERATION

3.7.6.3 Two independent control room air conditioning units shall be OPERABLE.

APPLICABILITY*: MODES 1, 2, 3, and 4.

ACTION:

a. With one control room air conditioning unit inoperable, restore the inoperable unit to OPERABLE status within 7 days or be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With two control room air conditioning units inoperable, return one unit to an OPERABLE status within 1 hour or be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.6.3 Each control room air conditioning unit shall be demonstrated OPERABLE:

a. At least once per 12 hours by verifying that the operating control room air conditioning unit is maintaining average control room air temperature less than or equal to 80°F.

b. At least quarterly, if not performed within the last quarter, by verifying that each control room air conditioning unit starts and operates for at least 15 minutes.

*During movement of irradiated fuel assemblies, TS 3.7.6.4 is also applicable.

PLANT SYSTEMS

~~3/4.7.6.4~~ CONTROL ROOM AIR TEMPERATURE - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.7.6.4 Two independent control room air conditioning units shall be OPERABLE.

APPLICABILITY: MODES 5, 6, and during movement of irradiated fuel assemblies.

ACTION:

a. With one control room air conditioning unit inoperable, restore the inoperable system to OPERABLE status within 7 days or initiate and maintain operation of the remaining OPERABLE control room air conditioning unit.

b. With both control room air conditioning units inoperable, or with the OPERABLE control room air conditioning unit, required to be in operation by ACTION a, not capable of being powered by an OPERABLE emergency power source, suspend all operations involving CORE ALTERATIONS and movement of irradiated fuel assemblies.

SURVEILLANCE REQUIREMENTS

4.7.6.4 The control room air conditioning units shall be demonstrated OPERABLE per the Surveillance Requirements of 4.7.6.3.

PLANT SYSTEMS

~~3/4.7.6.5~~ CONTROL ROOM ISOLATION AND PRESSURIZATION

LIMITING CONDITION FOR OPERATION

3.7.6.5 The control room envelope isolation and pressurization boundaries shall be OPERABLE.

APPLICABILITY: All MODES and during movement of irradiated fuel assemblies.

ACTION:

a. With either control room envelope isolation valve in a normal outside air flow path inoperable, maintain at least one isolation valve in the flowpath OPERABLE, and either restore the inoperable valve to OPERABLE status with 7 days or isolate the affected flow path within the following 6 hours.

b. With any Control Room Emergency Filter Outside Air Intake valve(s) inoperable, maintain at least one of the series isolation valves in a flowpath OPERABLE, and either restore the inoperable valve(s) to OPERABLE status within 7 days or isolate the affected flow path within the following 6 hours.

c. With more than one Control Room Emergency Filter Outside Air Intake flow path inoperable, maintain at least one flow path per intake operable and restore an additional flow path to operable status within 7 days or, be in HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours.

d. With the control room envelope inoperable as a result of causes other than those addressed by ACTION (a), (b), or (c) above:

   1. Within 1 hour and at least once per 12 hours thereafter while the control room envelope is inoperable, verify that the Emergency Breathing Airbanks pressure is greater than or equal to 1800 psig.

   2. MODES 1-4:

      a. If the cause of control room envelope inoperability is due to a known breach in the envelope of less than or equal to one square foot total area or the breach is associated with a permanent sealing mechanism (e.g., blocking open or removing a door) then operation may continue for up to 7 days after the control room envelope is declared inoperable. Otherwise, be in HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours.

~~TABLE 4.8-1~~

~~DIESEL GENERATOR TEST SCHEDULE~~

~~Not Used~~

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Pages 3/4 11-18 and 11-19 have been deleted.

DESIGN FEATURES

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The reactor core shall contain 217 fuel assemblies with each fuel assembly containing a maximum of 236 fuel rods clad with Zircaloy-4. Each fuel rod shall have a nominal active fuel length of 150 inches and contain a nominal total weight of 1830 grams uranium. The initial core loading shall have a maximum enrichment of 2.91 weight percent U-235. Reload fuel shall be similar in physical design to the initial core loading. Assemblies shall have a maximum nominal enrichment of 5.0 weight percent U-235 and meet the final storage requirements described in Section 5.6.

CONTROL ELEMENT ASSEMBLIES

5.3.2 The reactor core shall contain 87 control element assemblies.

5.4 ~~REACTOR COOLANT SYSTEM~~ Not Used

~~DESIGN PRESSURE AND TEMPERATURE~~

~~5.4.1 The Reactor Coolant System is designed and shall be maintained:~~

~~a. In accordance with the code requirements specified in Section 5.2 of the FSAR with allowance for normal degradation pursuant of the applicable Surveillance Requirements~~

~~b. For a pressure of 2500 psia, and~~

~~c. For a temperature of 650°F, except for the pressurizer and surge line which is 700°F.~~

~~VOLUME~~

~~5.4.2 The total water and steam volume of the reactor coolant system is 11,800 +600/-0 cubic feet at a nominal T_{avg} of 582/1°F.~~

5.5 METEOROLOGICAL TOWERS LOCATION

5.5.1 The primary and backup meteorological towers shall be located as shown on Figure 5.1-1.

DESIGN FEATURES

5.6 FUEL STORAGE

CRITICALITY

5.6.1 The spent fuel storage racks are designed and shall be maintained with:

a. A normal k_{eff} of less than or equal to 0 95 when flooded with unborated water, which includes a conservative allowance for uncertainties.

b. A nominal 10.185 inch center-to-center distance between fuel assemblies placed in Region 1 (cask storage pit) spent fuel storage racks.

c. A nominal 8.692 inch center-to-center distance between fuel assemblies in the Region 2 (spent fuel pool and refueling canal) racks, except for the four southern-most racks in the spent fuel pool which have an increased N-S center-to-center nominal distance of 8.892 inches.

d. New or partially spent fuel assemblies may be allowed unrestricted storage in Region 1 racks.

e. New fuel assemblies may be stored in the Region 2 racks provided that they are stored in a "checkerboard pattern" as illustrated in Figure 5.6-1.

f. Partially spent fuel assemblies with a discharge burnup in the "acceptable range" of Figure 5.6-2 may be allowed unrestricted storage in the Region 2 racks.

g Partially spent fuel assemblies with a discharge burnup in the "unacceptable range" of Figure 5.6-2 may be stored in the Region 2 racks provided that they are stored in a "checkerboard pattern", as illustrated in Figure 5.6-1, with spent fuel in the "acceptable range" of Figure 5.6-3.

5.6.2 The k_{eff} for new fuel stored in the new fuel storage racks shall be less than or equal to 0.95 when flooded with unborated water and shall not exceed 0.98 when aqueous foam moderation is assumed.

DRAINAGE

5.6.3 The spent fuel pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation +40.0 MSL. When fuel is being stored in the cask storage pit and/or the refueling canal, these areas will also be maintained at +40.0 MSL.

CAPACITY

5.6.4 The spent fuel pool is designed and shall be maintained with a storage capacity limited to no more than 1849 fuel assemblies in the main pool, 255 fuel assemblies in the cask storage pit and after permanent plant shutdown 294 fuel assemblies in the refueling canal. The heat load from spent fuel stored in the refueling canal racks shall not exceed 1.72 X 10E6 BTU/Hr. Fuel shall not be stored in the spent fuel racks in the cask storage pit or the refueling canal unless all of the racks are installed in each respective area per the design.

5.7 COMPONENT CYCLIC OR TRANSIENT LIMITS

5.7 1 The components identified in Table 5 7-1 are designed and shall be maintained within the cyclic or transient limits of Table 5.7-1.

UNIT STAFF (Continued)

4. Except during extended shutdown periods, the use of overtime shall be considered on an individual basis and not for the entire staff on a shift.

   Any deviation from the above guidelines shall be authorized by the General Manager Plant Operations, or designee, or higher levels of management, in accordance with established procedures and with documentation of the basis for granting the deviation. Controls shall be included in the procedures such that individual overtime will be reviewed monthly by the General Manager Plant Operations or his designee to assure that excessive hours have not been assigned. Routine deviation from the above guidelines is not authorized.

f. The Operations Manager or the ~~Operations Superintendent~~ shall hold a senior reactor operator license.

Assistant Operations Manager (shift)

TABLE 6.2-1

MINIMUM SHIFT CREW COMPOSITION

POSITION	NUMBER OF INDIVIDUALS REQUIRED TO FILL POSITION	
	MODE 1, 2, 3, OR 4	MODE 5 OR 6
~~SS~~ SM	1*	1
SRO	1*	None
RO	2	1
AO	2	1
STA	1*	None

~~SS~~ SM - Shift ~~Superintendent~~ Manager with a Senior Operator License
SRO - Individual with a Senior Operator License
RO - Individual with an Operator License
AO - Auxiliary Operator
STA - Shift Technical Advisor

Except for the Shift ~~Superintendent~~ Manager the shift crew composition may be one less than the minimum requirements of Table 6.2-1 for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements of Table 6.2-1. This provision does not permit any shift crew position to be unmanned upon shift change due to an oncoming shift crewman being late or absent.

During any absence of the Shift ~~Superintendent~~ Manager from the control room while the unit is in MODE 1, 2, 3 or 4, an individual (other than the Shift Technical Advisor) with a valid Senior Operator license shall be designated to assume the control room command function. During any absence of the Shift ~~Superintendent~~ Manager from the control room while the unit is in MODE 5 or 6, an individual with a valid Senior Operator or Operator license shall be designated to assume the control room command function.

*An individual with SRO/STA qualifications can satisfy the ~~SS~~ SM/STA or SRO/STA position requirements simultaneously.

6.2.3 Not Used

6.2.4 SHIFT TECHNICAL ADVISOR

6.2.4.1 The Shift Technical Advisor shall provide advisory technical support to the Shift ~~Superintendent~~* MANAGER in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the unit. The STA shall meet the requirements of either Option 1 or 2 as shown below:

a. Option 1 - Combined SRO/STA Position. This option is satisfied by assigning an individual with the following qualifications to each operating shift crew as one of the SRO's required by 10 CFR 50.54(m) (2) (i):

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*Not responsible for sign-off function.

Pages 6-9
through
page 6-~~12~~ 13
not used

6.6 REPORTABLE EVENT ACTION

6.6.1 The following actions shall be taken for REPORTABLE EVENTS:

a. The Commission shall be notified and a report submitted pursuant to the requirements of Section 50.73 to 10 CFR Part 50, and

6.7 SAFETY LIMIT VIOLATION

6.7.1 The following actions shall be taken in the event a Safety Limit is violated:

~~SAFETY LIMIT VIOLATION (Continued)~~

~~a. The NRC Operations Center shall be notified by telephone as soon as possible and in all cases within 1 hour. The Vice President Operations and the SRC shall be notified within 24 hours.~~

~~b. A Safety Limit Violation Report shall be prepared. This report shall describe (1) applicable circumstances preceding the violation, (2) effects of the violation upon facility components, systems, or structures, and (3) corrective action taken to prevent recurrence.~~

~~c. The Safety Limit Violation Report shall be submitted to the Commission, and the Vice President Operations within 14 days of the violation.~~

~~d. Critical operation of the unit shall not be resumed until authorized by the Commission.~~

6.6 Not Used

6.7 Not Used

6.8 PROCEDURES AND PROGRAMS

6.8.1 Written procedures shall be established, implemented and maintained covering the activities referenced below:

a. The applicable procedures recommended in Appendix A of Regulatory Guide 1.33, Revision 2, February 1978 and Emergency Operating Procedures required to implement the requirements of NUREG-0737 and NUREG-0737, Supplement 1, as stated in Generic Letter 82-33.
b. Refueling operations.
c. Surveillance and test activities of safety-related equipment.
d. Not used
e. Not used.
f. Fire Protection Program implementation.
g. Modification of Core Protection Calculator (CPC) Addressable Constants, including independent verification of modified constants.

NOTES:

(1) Modification to the CPC addressable constants based on information obtained through the Plant Computer - CPC data link shall not be made without prior approval of the ~~PORC~~. On-Site Safety Review Committee

(2) Modifications to the CPC software (including algorithm changes and changes in fuel cycle specific data) shall be performed in accordance with the most recent version of CEN-39(A)-P, "CPC Protection Algorithm Software Change Procedure," that has been determined to be applicable to the facility. Additions or deletions to CPC Addressable Constants or changes to Addressable Constant software limits values shall not be implemented without prior NRC approval.

h. Administrative procedures implementing the overtime guidelines of Specification 6.2.2e., including provisions for documentation of deviations.

i. PROCESS CONTROL PROGRAM implementation.

6.9 REPORTING REQUIREMENTS

ROUTINE REPORTS

6.9.1 ~~In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the~~ T following reports shall be submitted ~~to the Regional Administrator of the Regional Office of the NRC unless otherwise noted.~~ IN ACCORDANCE with 10 CFR 50.4.

STARTUP REPORT

6.9.1.1 A summary report of plant startup and power escalation testing shall be submitted following (1) receipt of an Operating License. (2) amendment to the license involving a planned increase in power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the plant.

6.9.1.2 The startup report shall address each of the tests identified in the FSAR and shall include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be included in this report.

6.9.1.3 Startup reports shall be submitted within (1) 90 days following completion of the startup test program, (2) 90 days following resumption or commencement of commercial power operation, or (3) 9 months following initial criticality, whichever is earliest. If the startup report does not cover all three events (i.e., initial criticality, completion of startup test program, and resumption or commencement of commercial operation), supplementary reports shall be submitted at least every 3 months until all three events have been completed.

ANNUAL REPORTS

6.9.1.4 Annual reports covering the activities of the unit as described below for the previous calendar year shall be submitted prior to March 1 of each year. The initial report shall be submitted prior to March 1 of the year following initial criticality.

The results of specific activity analysis in which the primary coolant exceed the limits of Specification 3.4.7 shall be submitted annually in accordance with the aforementioned time frame. The following information shall be included:

MONTHLY OPERATING REPORTS

6.9.1.6 Routine reports of operating statistics and shutdown experience, including documentation of all challenges to the safety valves, shall be submitted on a monthly basis~~, to the Director, Office of Resource Management, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, with a copy to the Regional Administrator of the Regional Office of the NRC,~~ no later than the 15th of each month following the calendar month covered by the report.

ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT

6.9.1.7 The Annual Radiological Environmental Operating Report covering the operation of the unit during the previous calendar year shall be submitted before May 1 of each year. The report shall include summaries, interpretations, and analysis of trends of the results of the Radiological Environmental Monitoring Program for the reporting period. The material provided shall be consistent with the objectives outlined in (1) the ODCM and (2) Sections IV.B.2, IV.B.3, and IV.C of Appendix I to 10 CFR Part 50.

ADMINISTRATIVE CONTROLS

CORE OPERATING LIMITS REPORT COLR (Continued)

6) "CESEC - Digital Simulation for a Combustion Engineering Nuclear Steam Supply System," CENPD-107, December 1981. (Methodology for Specification 3.1.1.1 and 3.1.1.2 for Shutdown Margins, 3.1.1.3 for MTC, 3.1.3.1 for Movable Control Assemblies - CEA Position, 3.1.3.6 for Regulating and group P CEA Insertion Limits, and 3.2.3 for Azimuthal Power Tilt).

7) "Qualification of Reactor Physics Methods for the Pressurized Water Reactors of the Entergy System," ENEAD-01-P, Revision 0. (Methodology for Specifications 3.1.1.1 and 3.1.1.2 for Shutdown Margins, 3.1.1.3 for MTC, 3.1.3.6 for Regulating and group P CEA Insertion Limits, 3.1.2.9 Boron Dilution (Calculation of CBC & IBW), and 3.9.1 Boron Concentration)

6.9.1.11.2 The core operating limits shall be determined such that all applicable limits (e.g , fuel thermal limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

6.9.1:11.3 The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance, for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted ~~to the Regional Administrator of the Regional Office of the NRC~~ within the time period specified for each report.

In Accordance with 10CFR 50.4

6.10 Not Used

Attachment 3

W3F1-2002-0107

Changes to Technical Specification Bases Pages
(For Information Only)

POWER DISTRIBUTION LIMITS

BASES

3/4.2.2 PLANAR RADIAL PEAKING FACTORS -F_{xy}

Limiting the values of the PLANAR RADIAL PEAKING FACTORS (F^c_{xy}) used in the COLSS and CPCs to values equal to or greater than the measured PLANAR RADIAL PEAKING FACTORS (F^m_{xy}) provides assurance that the limits calculated by COLSS and the CPCs remain valid. Data from the incore detectors are used for determining the measured PLANAR RADIAL PEAKING FACTORS. A minimum core power at 20% of RATED THERMAL POWER is assumed in determining the PLANAR RADIAL PEAKING FACTORS. The 20% RATED THERMAL POWER threshold is due to the neutron flux detector system being inaccurate below 20% core power. Core noise level at low power is too large to obtain usable detector readings. The periodic Surveillance Requirements for determining the measured PLANAR RADIAL PEAKING FACTORS provide assurance that the PLANAR RADIAL PEAKING FACTORS used in COLSS and the CPCs remain valid throughout the fuel cycle. Determining the measured PLANAR RADIAL PEAKING FACTORS after each fuel loading prior to exceeding 70% of RATED THERMAL POWER provides additional assurance that the core was properly loaded.

3/4.2.3 AZIMUTHAL POWER TILT - T_q

The limitations on the AZIMUTHAL POWER TILT are provided to ensure that design safety margins are maintained. The LCO requires the maximum azimuthal tilt during normal steady state power operation to be less than or equal to that specified in the COLR. With AZIMUTHAL POWER TILT greater than the limit specified in the COLR, operation is restricted to only those conditions required to identify the cause of the tilt. However, Action item b.2 allows 24 hours to restore the tilt to less than or equal to the limit specified in the COLR following a CEA misalignment event (i.e., CEA drop). A CEA misalignment event causes an asymmetric core power generation and an increase in xenon concentration in the vicinity of the dropped rod. This event may cause the azimuthal tilt to exceed the limit specified in the COLR. The 2 hour action time to reduce core power is not sufficient to recover from the xenon transient. The 24 hour period allows for correction of the misaligned CEA and allows time for the xenon redistribution effects to dampen out due to radioactive decay and absorption. The reduction in xenon concentration (which is aided by operation at full power) will in turn reduce the tilt below the COLR limit.

The 24 hour period is applicable only to a CEA misalignment where the cause of the tilt has been identified. It is based on the time required or the expected xenon transient to dampen out. All other conditions (not due to a CEA misalignment) where the azimuthal tilt exceeds the limit specified in the COLR require action within the specified 2 hours.

The tilt is normally calculated by COLSS. A minimum core power of 20% of RATED THERMAL POWER is assumed by the CPCs in its input to COLSS for calculation of AZIMUTHAL POWER TILT. The 20% RATED THERMAL POWER threshold is due to the neutron flux detector system being inaccurate below 20% core power. Core noise level at low power is too large to obtain usable detector readings. The Surveillance Requirements specified when COLSS is out of service provide an acceptable means of detecting the presence of a steady-state tilt. It is necessary to explicitly account for power asymmetries in the COLSS and CPCs because the radial peaking factors used in the core power distribution calculations are based on an untilted power distribution.

3/4.3.3.6 ACCIDENT MONITORING INSTRUMENTATION

The OPERABILITY of the accident monitoring instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess these variables following an accident. The availability of accident monitoring instrumentation is important so that responses to corrective actions can be observed and the need for, and magnitude of, further actions can be determined. These essential instruments are identified by plant specific documents addressing the recommendations of Regulatory Guide 1.97, as required by Supplement 1 to NUREG-0737, "TMI Action Items." Table 3.3.10 includes most of the plant's RG 1.97 Type A and Category 1 variables. The remaining Type A/Category 1 variables are included in their respective specifications. Type A variables are included in this LCO because they provide the primary information required to permit the control room operator to take specific manually controlled actions, for which no automatic control is provided, that are required for safety systems to accomplish their safety functions for Design Basis Accidents (DBAs). Category 1 variables are the key variables deemed risk significant because they are needed to: (1) Determine whether other systems important to safety are performing their intended functions; (2) Provide information to the operators that will enable them to determine the potential for causing a gross breach of the barriers to radioactivity release; and (3) Provide information regarding the release of radioactive materials to allow for early indication of the need to initiate action necessary to protect the public as well as to obtain an estimate of the magnitude of any impending threat.

With the number of OPERABLE accident monitoring channels less than the Required Number of Channels shown in Table 3.3-10, the inoperable channel should be restored to OPERABLE status within 30 days. The 30 day Completion Time is based on operating experience and takes into account the remaining OPERABLE channel (or in the case of a Function that has only one required channel, other non-Regulatory Guide 1.97 instrument channels to monitor the Function), the passive nature of the instrument (no critical automatic action is assumed to occur from these instruments), and the low probability of an event requiring accident monitoring instrumentation during this interval. If the 30 day AOT is not met, a Special Report approved by ~~PORC~~ is required to be submitted to the NRC within the following 14 days. This report discusses the results of the root cause evaluation of the inoperability and identifies proposed restorative Actions. This Action is appropriate in lieu of a shutdown requirement, given the likelihood of plant conditions that would require information provided by this instrumentation. Also, alternative Actions are identified before a loss of functional capability condition occurs.

OSRC

With the number of OPERABLE accident monitoring channels less than the Minimum Channels OPERABLE requirements of Table 3.3-10; at least one of the inoperable channels should be restored to OPERABLE status within 7 days. The Completion Time of 7 days is based on the relatively low probability of an event requiring PAM instrumentation operation and the availability of alternate means to obtain the required information.

Continuous operation with less than the Minimum Channels OPERABLE requirements is not acceptable because the alternate indications may not fully meet all performance qualification requirements applied to the accident

PLANT SYSTEMS

BASES

3/4 7.5 FLOOD PROTECTION

The limitation on flood protection ensures that facility protective actions will be taken in the event of flood conditions. The limit of elevation 27.0 ft Mean Sea Level is based on the maximum elevation at which the levee provides protection, the nuclear plant island structure provides protection to safety-related equipment up to elevation +30 ft Mean Sea Level.

3/4.7.6 CONTROL ROOM AIR CONDITIONING SYSTEM

~~3/4 7.6.1 and 3/4.7.6.2 CONTROL ROOM EMERGENCY AIR FILTRATION SYSTEM~~

During an emergency, both S-8 units are started to provide filtration and adsorption of outside air and control room envelope recirculated air (reference: FSAR 6.4.3.3). Dosages received after a full power design basis LOCA were calculated to be orders of magnitude higher than other accidents involving radiation releases to the environment (reference: FSAR Tables 15.6-18, 15.7-2, 15.7-4, 15.7-5, 15.7-7).

Acceptable removal efficiency is shown by a methyl iodide penetration of less than 0.5% when tests are performed in accordance with ASTM D3803-1989, "Standard Test Method for Nuclear-Grade Activated Carbon," at a temperature of 30°C and a relative humidity of 70%. The penetration acceptance criterion is determined by the following equation:

$$\text{Allowable Penetration} = \frac{[100\% - \text{methyl iodide efficiency for charcoal credited in accident analysis}]}{\text{safety factor of 2}}$$

Applying a safety factor of 2 is acceptable because ASTM D3803-1989 is a more accurate and demanding test than older tests.

The OPERABILITY of this system and control room design provisions are based on limiting the radiation exposure to personnel occupying the control room to 5 rem or less whole body, or its equivalent. This limitation is consistent with the requirements of General Design Criterion 19 of Appendix A, 10 CFR Part 50.

The ACTION to suspend all operations involving movement of irradiated fuel assemblies shall not preclude completion of movement to a safe conservative position.

Operation of the system with the heaters on for at least 10 hours continuous over a 31-day period is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters. Obtaining and analyzing charcoal samples after 720 hours of adsorber operation (since the last sample and analysis) ensures that the adsorber maintains the efficiency assumed in the safety analysis and is consistent with Regulatory Guide 1.52 and ASTM D3803-1989.

PLANT SYSTEMS

BASES

3/4.7.8 SNUBBERS

All snubbers are required OPERABLE to ensure that the structural integrity of the reactor coolant system and all other safety-related systems is maintained during and following a seismic or other event initiating dynamic loads. Snubbers excluded from this inspection program are those installed on nonsafety-related systems and then only if their failure or failure of the system on which they are installed, would have no adverse effect on any safety-related system.

Snubbers are classified and grouped by design and manufacturer but not by size. For example, mechanical snubbers utilizing the same design features of the 2-kip, 10-kip, and 100-kip capacity manufactured by Company "A" are of the same type. The same design mechanical snubbers manufactured by Company "B" for the purposes of this Technical Specification would be of a different type, as would hydraulic snubbers from either manufacturer.

Onsite

A list of individual snubbers with detailed information of snubber location and size and of system affected shall be available at the plant in accordance with Section 50.71(c) of 10 CFR Part 50. The accessibility of each snubber shall be determined and approved by the ~~Plant Operations~~ Review Committee. The determination shall be based upon the existing radiation levels and the expected time to perform a visual inspection in each snubber location as well as other factors associated with accessibility during plant operations (e.g., temperature, atmosphere, location, etc.), and the recommendations of Regulatory Guides 8.8 and 8.10. The addition or deletion of any hydraulic or mechanical snubber shall be made in accordance with Section 50.59 of 10 CFR Part 50.

Safety

Guidance for visual inspection is provided in NRC Generic Letter 90-09. A visual inspection is the observation of the condition of installed snubbers to identify those that are damaged, degraded, or inoperable as caused by physical means, leakage, corrosion, or environmental exposure. The functional testing program provides a 95 percent confidence level that 90 to 100 percent of the snubbers will operate within the specified acceptance limits. The performance of visual examinations is a separate process that complements the functional testing program and provides additional confidence in snubber operability.